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Abstract

CRITICAL HEAT FLUX CONSIDERATIONS IN THE THERMAL AND HYDRAULIC DESIGN OF WATER COOLED NUCLEAR REACTORS

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The critical heat flux condition is one of the most important thermal and hydraulic limits in water cooled nuclear reactors. Its occurrence is associated with a rapid deterioration of the heat transfer process and is accompanied by temperature oscillations and a sharp temperature rise which can be detrimental to the integrity of fuel elements. This paper is concerned with three aspects of the critical heat flux condition in water cooled nuclear reactors. They are (1) the experimental measurement of the critical heat flux condition, (2) its analytical or empirical prediction, and (3) the application of these measurements or predictions to the design of nuclear reactors.

Recent measurements in simplified geometries such as circular, rectangular and annular are reviewed and analyzed. The test data cover a wide range of variables, i.e., pressure of 500 to 1500 psia, flow rate of 0.1 to 13 x 10^6 lb/hr-ft^2, hydraulic diameter of 0.18 to 1.5 inch, heated length of 24 to 262 inches, and various modes of heating. The major parameters and their effects upon the critical heat flux condition are identified and described. Comparison of the results reported for the various simplified geometries and reference to previously obtained data are used to single out the remaining important sources of uncertainties, namely the effects of unheated surface area, heat flux distribution, small hydraulic diameter, and special loop test characteristics.

Test results with a four, seven, nine, twelve, and nineteen rod assembly are covered. The effects of flow rate, system pressure, rod to rod spacing, uneven heating distribution, and heated length are identified and discussed. The important role of rod to rod spacing devices and their potential helpful or harmful contributions are emphasized. Comparison of the multirod data

with corresponding simplified geometry results shows that the internally heated annulus and the circular geometry act respectively as a lower and upper limit for the multirod results.

Recent empirical correlations and analytical predictions of critical heat flux conditions are next described. The correlations of Gambill, Green, Macbeth, and Tong, and the analytical predictions of Becker, Collier, Grace, Isbin, and Tippets are outlined. Major discrepancies among the various proposed relations are noted. Most correlations are found to apply only over a limited range of test geometry and conditions.

The application of the above experimental results and predictions to the design of nuclear reactors is discussed. First, other sources of uncertainties in design or operating calculations such as power, flow, and coolant enthalpy distribution are briefly reviewed. Next, the required design margins are evaluated in terms of the present state of knowledge of the critical heat flux condition and of the consequences of operation beyond it. Recent test data with a single and two-rod geometry are described. Temperature oscillation and film boiling coefficients prevailing beyond the critical heat flux condition are reported. A reactor test with a fuel rod operating beyond the critical heat flux is also described. These results show that under appropriate operating and design conditions the consequences of operation beyond the critical heat flux are predictable and not serious.